May 7, 1998

EA Nos. 98-150; 98-151; 98-152; 98-186

Mr. E. E. Fitzpatrick
Executive Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: NRC INSPECTION REPORT NO. 50-315/98009(DRS); 50-316/98009(DRS)

Dear Mr. Fitzpatrick:

On April 15, 1998, the NRC completed an inspection at your D. C. Cook, Units 1 and 2 reactor facilities. The purpose of this inspection was to determine the safety significance and regulatory impact of 34 concerns identified during the 1997 Architectural and Engineering (AE) inspection (50-315/97201; 50-316/97201). The enclosed report presents the results of this inspection.

Based on the results of this inspection, 15 apparent violations were identified and are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600.
These apparent violations are grouped into three programmatic areas: design control, 10 CFR 50.59 safety evaluations, and corrective actions.

Eight apparent violations of 10 CFR 50, Appendix B, Criterion III, "Design Control," were identified. Specifically, three engineering calculations were not adequately verified or checked. Two of these examples pertained to refueling water storage tank (RWST) level measurement biases and instrument measurement uncertainties not being accounted for in the development of the RWST low and low-low level setpoints. The third example involved the containment sump level post-accident instrument uncertainties. In addition, five examples were identified where the plant design basis was not correctly translated into specifications, drawings, procedures, and instructions. Three of these examples pertained to the containment recirculation sump design basis. For example, the sump water volume requirement lacked documentation to demonstrate that sufficient water was available to prevent air entrainment in the emergency core cooling and containment spray pumps, the sump roof ¾ inch vent hole installation commitment was not incorporated into the safety analysis report, and the sump ¼ inch particulate retention requirement was not maintained. The fourth example involved a calculated component cooling water (CCW) flow value that exceeded the Updated Final Safety Analysis Report (UFSAR) design value. The fifth example involved an RWST Appendix R volume requirement that was not incorporated in a shutdown risk procedure. Collectively, these apparent violations represent a programmatic breakdown in the maintenance and control of facility design.

Six apparent violations of 10 CFR 50.59, “Changes, Tests and Experiments,” were identified where inadequate safety evaluations were performed. Several safety evaluations inadequately addressed system operations that were outside UFSAR stated design values. For example, the units were operated above the ultimate heat sink (lake) temperature value, the CCW system design temperature value was exceeded, the units were operated with less than the stated reactor coolant pump thermal barrier CCW flow value, and the spent fuel pool time-to-boil margin was reduced during a dual train CCW and emergency service water outage. In addition, the safety evaluation for changes to emergency operating procedure Nos. 01(02)-OHP 4023.ES-1.3, “Transfer to Cold Leg Recirculation,” failed to identify that the changes created a single failure vulnerability. Collectively, these apparent violations represent a programmatic breakdown in the implementation of the safety evaluation process to identify unreviewed safety questions.

One apparent violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” was identified. The control room temperature evaluation calculation identified in 1990 that high emergency service water (lake) temperatures would reduce the qualified life of control room components needed for plant shutdown, however, this condition had not been appropriately evaluated.

No Notice of Violation is presently being issued for these apparent violations. In addition, be advised that the number and characterization of the apparent violations described in the enclosed inspection report may change as a result of further NRC review.

An open predecisional enforcement conference to discuss these apparent violations has been scheduled for May 20, 1998. The decision to hold a predecisional enforcement conference does not mean that the NRC has determined that violations occurred or that enforcement action will be taken. This conference will be held to obtain information to enable the NRC to make an enforcement decision, such as a common understanding of the facts, root causes, missed opportunities to identify the apparent violations sooner, corrective actions, significance of the issues, and the need for lasting and effective corrective action. In addition, this is an opportunity for you to provide any information concerning your perspectives on: 1) the severity of the violations, 2) the application of the factors that the NRC considers when it determines the amount of a civil penalty that may be assessed in accordance with Section VI.B.2 of the Enforcement Policy, and 3) any other application of the Enforcement Policy to this case, including the exercise of discretion in accordance with Section VII.

The remaining nineteen AE team identified unresolved items appear to be violations of NRC requirements. However, it appears that the actions necessary to correct these issues would be similar to planned or completed actions that you will discuss during the scheduled predecisional enforcement conference. Therefore, no Notice of Violation is presently being issued for these violations.

You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding these apparent violations is required at this time.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

original /s/ J. A. Grobe
John A. Grobe, Director
Division of Reactor Safety

Docket Nos. 50-315, 50-316
License Nos. DPR-58, DPR-74

Enclosure: Inspection Report Nos. 50-315/98009(DRS);
50-316/98009(DRS)

cc w/encl: John Sampson, Site Vice President
A. A. Blind, Vice President Nuclear Engineering
Douglas Cooper, Plant Manager
Richard Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division, MI Department of State Police

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No Notice of Violation is presently being issued for these apparent violations. In addition, be advised that the number and characterization of the apparent violations described in the enclosed inspection report may change as a result of further NRC review.

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Sincerely,

John A. Grobe, Director
Division of Reactor Safety

Docket Nos. 50-315, 50-316
License Nos. DPR-58, DPR-74

Enclosure: Inspection Report Nos. 50-315/98009(DRS);
50-316/98009(DRS)

cc w/encl:
John Sampson, Site Vice President
A. A. Blind, Vice President Nuclear Engineering
Douglas Cooper, Plant Manager
Richard Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division, MI Department of State Police

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos: License Nos:	50-315; 50-316 DPR-58; DPR-74
Report Nos:	50-315/98009(DRS); 50-316/98009(DRS)
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	April 14 through 15, 1998.
Inspector:	D. Butler, Reactor Engineer
Approved by:	R. N. Gardner, Chief Engineering Specialists Branch 2 Division of Reactor Safety

EXECUTIVE SUMMARY

D. C. Cook, Units 1 and 2
NRC Inspection Report Nos. 50-315/98009(DRS); 50-316/98009(DRS)

The purpose of this inspection was to determine the safety significance and regulatory impact of 34 concerns identified during the 1997 Architectural and Engineering (AE) inspection (50-315/97201; 50-316/97201). The following observations were made:

<u>Design Control</u>

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to verify or check the adequacy of Engineering Control Procedure (ECP) calculation Nos. 1-RCP-09 and 2-RCP-09, "RWST Level." Specifically, the suction pipe entrance head losses and Bernoulli velocity head losses were not included in the uncertainty analysis. (Section E8.1)

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to verify or check the adequacy of ECP Nos. 1-CG-39 and 2-CG-39, "Refueling Water Storage Tank Level." Specifically, vortexing (air entrainment) was not addressed when the RWST low-low level setpoint was developed. (Section E8.3)

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to verify or check the adequacy of ECP Nos. 1-2-N3-01, "CNTMT Sump Water Level Indication," 1-RPC-14 and 2-RPC-14, "Containment/Containment Sump Level." Specifically, post-accident containment environment effects were not incorporated in the uncertainty analysis. (Section E8.5)

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to correctly translate containment water inventory requirements into specifications, drawings, procedures, and instructions. Specifically, it was not demonstrated that sufficient water could be recovered during a design basis accident to prevent pump vortexing. (Section E8.6)

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to correctly translate ¼ inch containment sump particulate retention requirements into specifications, drawings, procedures, and instructions. Specifically, the containment sump screen sections contained ½ inch gaps and the ¾ inch sump roof vent holes were not covered with screening material. (Section E8.8)

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to correctly translate CCW heat exchanger design flow into specifications, drawings, procedures, and instructions. Specifically, the cooldown analysis and operating procedures used a CCW flow that exceeded the UFSAR design value. (Section E8.14)

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to correctly translate RWST Appendix R inventory requirements into specifications, drawings, procedures, and instructions. Specifically, calculation No. TH-90-02, "RCS Volume Make-up Required After Appendix R Fire," RWST volume requirements were not incorporated into procedure No. PMP-4100, "Plant Shutdown Safety and Risk Management." (Section E8.18)

- An apparent violation of 10 CFR 50, Appendix B, Criterion III, was identified pertaining to the failure to correctly translate the ¾ inch recirculation sump roof vent hole design into specifications, drawings, procedures, and instructions. Specifically, the vent holes were plugged without verifying their design basis. (Section E8.31)

<u>50.59 Safety Evaluations</u>

- An apparent violation of 10 CFR 50.59, "Changes, Tests, and Experiments," was identified for not fully analyzing unit operation above UFSAR Tables 6.3-2 and 9.5-3 ESW 76°F ultimate heat sink (lake) design temperature. Specifically, the units were operated in 1988 for 22 continuous days at an average lake temperature of 81°F. (Section E8.28)

- An apparent violation of 10 CFR 50.59, "Changes, Tests, and Experiments," was identified for not considering the loss of spent fuel pool (SFP) cooling during a design basis accident. Specifically, the safety evaluations for the Unit 2 dual train CCW/ESW outage did not address the reduction in SFP time-to-boil if the Unit 1 CCW flow isolated due to a Unit 1 design basis accident. (Section E8.29)

- An apparent violation of 10 CFR 50.59, "Changes, Tests, and Experiments," was identified for creating a single failure vulnerability in a procedure revision to ES-1.3, "Transfer to Cold Leg Recirculation." Specifically, Revision 2 to ES-1.3 piggy-backed all high head injection pumps onto one residual heat removal pump. (Section E8.30)

- An apparent violation of 10 CFR 50.59, "Changes, Tests, and Experiments," was identified for not performing a safety evaluation for unit operation with CCW temperatures in excess of the 95°F UFSAR Table 9.5-3 design value. (Section E8.32)

- An apparent violation of 10 CFR 50.59, "Changes, Tests, and Experiments," was identified for not performing a safety evaluation for unit operation with reactor coolant pump thermal barrier flow less than the 35 gpm UFSAR Table 9.5-2 design value. (Section E8.33)

- An apparent violation of 10 CFR 50.59, "Changes, Tests, and Experiments," was identified for not performing a safety evaluation for residual heat removal operation without automatic overpressure protection as described in UFSAR Section 9.3, "Residual Heat Removal System." (Section E8.34)

Corrective Actions

- An apparent violation of 10 CFR 50, Appendix B, Criterion XVI, was identified pertaining to not promptly correcting an identified condition adverse to quality. Specifically, calculation No. DCCHV12CR11N, “Control Room Temperature Evaluation,” identified in 1990 that control room equipment/component life could be reduced to 12 hours if the ESW temperature reached 87.5°F. Adequate documentation to demonstrate control room equipment shutdown capability at elevated temperatures could not be located. (Section E8.12)

Report Details

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Unresolved Item 50-315/97201-01; 50-316/97201-01: Refueling water storage tank (RWST) level setpoint error due to flow-induced effects.

The Architectural and Engineering (AE) team noted that the RWST level instruments pressure taps were located on the RWST suction pipe for the emergency core cooling system (ECCS) pumps and containment spray (CTS) pumps. The maximum flow rate expected during a design basis accident was about 17,800 gpm. At this flow rate, the suction pipe entrance head loss and Bernoulli velocity head loss would cause the indicated RWST tank level to be lower than the actual tank level. Engineering Control Procedure (ECP) Instrument uncertainty calculation No. 1-RCP-09, dated November 1, 1994, "RWST Level," did not include these head losses. As a consequence, an indicated lower tank level could affect ECCS and CTS pump suction transfers from the RWST to the containment recirculation sump during a design basis accident. This could lead to a premature transfer to the sump causing ECCS and CTS pump loss due to vortexing (air entrainment) and/or loss of net positive suction head (NPSH) from insufficient sump water level. These errors also affected ECP No. 2-RCP-09. The failure to verify or check the adequacy of ECP Nos. 1-RCP-09 and 2-RCP-09 is considered an apparent violation (EEI 50-315/98009-01; EEI 50-316/98009-01) of 10 CFR 50, Appendix B, Criterion III, "Design Control."

E8.2 (Closed) Unresolved Item 50-315/97201-02; 50-316/97201-02: RWST level instrument loop uncertainties were not accounted for in the Technical Specification (TS) volume verification surveillance.

The AE team was concerned that procedure Nos. 01(02)-OHP 4030.STP.030, "Daily and Shift Surveillance Checks," Revision 25(23), verified the RWST volume to be greater than the TS required 350,000 gallons (89%) without accounting for instrument loop uncertainties. This item is considered an unresolved item (URI 50-315/98009-02; URI 50-316/98009-02) pending further NRC review.

E8.3 (Closed) Unresolved Item 50-315/97201-03; 50-316/97201-03: RWST low-low level residual heat removal (RHR) and CTS pump automatic trip did not include all instrument loop uncertainties.

The AE team was concerned that vortexing (air entrainment) could take place when the ECCS and CTS pumps were aligned to the RWST. The low-low level setpoint as described in ECP No. 1-CG-39, dated October 24,1994, "Refueling Water Storage Tank Level," was set at 9 inches above the RWST discharge pipe. However, the ECP did not address the potential for vortexing. The licensee determined that vortexing could occur 12 inches above the discharge pipe. As a consequence, the potential existed to damage the ECCS and CTS pumps due to vortexing. These errors also affected ECP No. 2-CG-39. The failure to verify or check the adequacy of ECP Nos. 1-CG-39 and 2-CG-39 is considered an apparent violation (EEI 50-315/98009-03; EEI 50-316/98009-03) of 10 CFR 50, Appendix B, Criterion III, "Design Control."

E8.4 (Closed) Unresolved Item 50-315/97201-04; 50-316/97201-04: Potential for vortexing (air entrainment) was not adequately addressed in a timely manner.

The RWST level biases due to velocity effects were identified during the NRC Systems Based Instrument and Control Inspection (50-315/93012; 50-316/93012). The licensee determined that from an RWST inventory point of view the bias effects were conservative since the indicated level was lower than the actual level. However, the AE team believed that the overall vortexing effects could have been identified earlier. This is considered an unresolved item (URI 50-315/98009-04; URI 50-316/98009-04) pending further NRC review.

E8.5 (Closed) Unresolved Item 50-315/97201-05; 50-316/97201-05: Containment sump level instrument loops did not include post-accident uncertainties.

Calculation ECP Nos. 1-2-N3-01, dated March 16, 1994, “Redundant CNTMT Water and CNTMT Sump Water Level Indication,” 1-RPC-14, dated May 17, 1994, “Containment/Containment Sump Level,” 2-RPC-14, dated May 17, 1994, “Containment/Containment Sump Level,” did not account for the loop uncertainty impact on post-accident containment levels, did not include considerations for RHR and CTS pumps NPSH requirements, and did not account for pump vortexing (air entrainment). As a consequence, this could impact ECCS and CTS pumps during operator manual transfer from the RWST to the containment sump when implementing emergency operating procedure (EOP) Nos. 01(02)-OHP 4023. ES-1.3, “Transfer to Cold Leg Recirculation.” The failure to verify or check the adequacy of ECP Nos. 1-2-N3-01, 1-RPC-14, and 2-RPC-14 is considered an apparent violation (EEI 50-315/98009-05; EEI 50-316/98009-05) of 10 CFR 50, Appendix B, Criterion III, “Design Control.”

E8.6 (Closed) Unresolved Item 50-315/97201-06; 50-316/97201-06: Containment recirculation sump design basis water volume requirement was not maintained.

As of September 12, 1997, engineering reviews evaluating design basis accident flow diversions into the inactive portions of the containment sump could not be located. As a consequence, it was not known if sufficient water could be recovered during a design basis accident to prevent ECCS and CTS pump vortexing (air entrainment). This could jeopardize long term pump operation. The failure to correctly translate containment sump water inventory requirements into specifications, drawings, procedures, and instructions is considered an apparent violation (EEI 50-315/98009-06; EEI 50-316/98009-06) of 10 CFR 50, Appendix B, Criterion III, “Design Control.”

E8.7 (Closed) Unresolved Item 50-315/97201-07; 50-316/97201-07: The licensee’s definition for a “single active failure” was not consistent with the AE team’s interpretation.

The licensee indicated that their failure modes and effects analyses only had to postulate a "single active failure" as a failure-to-start. However, the AE team concluded that failure-to-run effects should also be analyzed. In response, the licensee contacted Westinghouse (W) and were informed that W also considered failure-to-run scenarios in their failure analyses. This is considered an unresolved item (URI 50-315/98009-07; URI 50-316/98009-07) pending NRC review of the licensee's failure modes and effects analyses.

E8.8 (Closed) Unresolved Item 50-315/97201-08; 50-316/97201-08: Recirculation sump screen edge gaps exceeded the containment ¼ inch particulate retention requirement.
The recirculation sump particulate retention requirement limited particle sizes to less than ¼ inch to prevent plugging of the ⅜ inch containment spray nozzles. Request for Change (RFC) No. DC-12-2361, completed July 9, 1979, "Modification to the Recirculation Sump," removed the sump horizontal perforated plate in the recirculation sump and installed a fine particulate screen behind the vertical grating at the sump entrance. However, the fine screen installation was deficient in that the individual screen section edges contained ½ inch gaps. In addition, ¾ inch containment sump roof vent holes had been installed without screens. As a consequence, a common mode failure of redundant CTS trains could occur. The failure to correctly translate containment sump particulate retention requirements into specifications, drawings, procedures, and instructions is considered an apparent violation (EEI 50-315/98009-08; EEI 50-316/98009-08) of 10 CFR 50, Appendix B, Criterion III, "Design Control."

E8.9 (Closed) Unresolved Item 50-97201-09; 50-316/97201-09: Untested ECCS backflow paths to the RWST during design basis accident recirculation.

Following a postulated loss of coolant accident (LOCA), with the ECCS operating in the recirculation mode, the ECCS piping located outside containment could provide an unmonitored and unfiltered leakage path back through the RWST. The RWST was vented to the atmosphere. Four of six valves were not leak tested. This is considered an unresolved item (URI 50-315/98009-09; URI 50-316/98009-09) pending NRC verification that the total back leakage was less than 10 gpm.

E8.10 (Closed) Unresolved Item 50-315/97201-10; 50-316/97201-10: Capability to cooldown the plant with one CCW train.

TS 3.0.3 stated, in part, that when a limiting condition for operation was not met, the plant was to be brought to a cold shutdown condition within 36 hours. The AE team determined that this requirement should be achievable with one CCW train. In response, the licensee provided the AE team calculation No. NEMP 960519AF, dated June 25, 1996, "CCW LOCA/Cooldown Analysis for the Unit 2 Uprating Program." However, the calculation used an emergency service water (ESW) temperature of 87.5°F and a CCW temperature of 120°F. Both temperature values exceeded UFSAR design values, 76°F and 95°F, respectively. This is considered an unresolved item (URI 50-315/98009-10; URI 50-316/98009-10) pending further NRC review.

E8.11 (Closed) Unresolved Item 50-315/97201-11; 50-316/97201-11: CCW heat exchanger type incorrectly modeled in the cooldown analysis.

The licensee identified that the CCW heat exchanger was modeled as a counter flow type rather that a single pass heat exchanger. This is considered an unresolved item (URI 50-

315/98009-11; URI 50-316/98009-11) pending NRC verification that sufficient margin exists in the cooldown analysis to accommodate the heat exchanger modeling error.

E8.12 (Closed) Unresolved Item 50-315/97201-12; 50-31697201-12: Identified decreases in control room equipment life due to high temperatures were not promptly corrected.

Calculation No. DCCHV12CR11N, dated June 22, 1990, “Control Room Temperature Evaluation,” determined that the minimum control room equipment/component life at an ESW temperature of 87.5°F was 12 hours. However, adequate documentation to demonstrate control room equipment shutdown capability at elevated temperatures could not be located. As a consequence, the decrease in control room equipment qualified life from 15,000 hours at an ESW temperature of 76°F to 12 hours at 87.5°F could impact plant shutdown with a loss of the nonsafety control room chillers during a design basis accident. The failure to promptly correct an identified condition adverse to quality is considered an apparent violation (EEI 50-315/98009-12; EEI 50-316/98009-12) of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.”

E8.13 (Closed) Unresolved Item 50-315/97201-13; 50-316/97201-13: Previously unanalyzed failure modes in the instrument air system.

The AE team evaluated safety related control room air-operated valve (AOV) failure modes. The valves were supplied from a common nonsafety air header and had been evaluated for a loss of control air. However, the failure analysis did not evaluate the potential for the control air regulators to fail high. This affected the 20 psig, 50 psig, and 85 psig air headers. This is considered an unresolved item (URI 50-315/98009-13; URI 50-316/98009-13) pending further NRC review.

Subsequently, the licensee modified the nonsafety air header in both units by installing relief valves downstream of each header control air regulator.

E8.14 (Closed) Unresolved Item 50-315/97201-14; 50-316/97201-14: CCW heat exchanger flow in excess of the UFSAR design value.

The UFSAR specified cooldown flow rate was about 8000 gpm (based on UFSAR Table 9.5-3 flow rate of 4.0×10^6 lbs/hour). Calculation No. SAE/FSE-C-AEP.AMP-0088, dated August 20, 1997, “D. C. Cook Units 1 & 2 RHR Cooldown Analysis for a JPO,” used a CCW flow rate of 4.35×10^6 lbs/hour which equates to about 8700 gpm through the shell side of the CCW heat exchanger. In addition, procedure Nos. 01(02)-OHP 4021.016.003, dated December 3, 1996, “Operation of the Component Cooling Water System During Reactor Startup and Normal Operation,” permitted CCW flows up to 9000 gpm. An engineering evaluation had not been performed to demonstrate the acceptability of this condition. As a consequence, the potential existed for the CCW heat exchangers to have increased vibration and/or to structurally fail. The failure to correctly translate the CCW heat exchanger design flow into specifications, drawings, procedures, and instructions is considered an apparent violation (EEI 50-315/98009-14; EEI 50-316/98009-14) of 10 CFR 50, Appendix B, Criterion III, “Design Control.”

Since the AE inspection, the licensee performed design change notice (DCN) No. 6944, dated January 28, 1998, “Safety Evaluation for UFSAR Changes to Add Maximum Flow Rates for Component Cooling Water Heat Exchanger and Letdown Heat Exchanger.” The

DCN indicated that the CCW heat exchanger could withstand an inlet and outlet bundle velocity up to 8 ft/sec. The maximum expected bundle velocity was 4.14 ft/sec at 9000 gpm. In addition, Tubular Exchangers Manufacturers Association Standards recommended that the heat exchanger tubes be supported at least every 52 inches. The CCW heat exchanger tubes were supported every 30 inches. This documentation supports that the CCW heat exchangers could withstand a 9000 gpm flow rate without a significant increase in flow induced vibration.

E8.15 (Closed) Unresolved Item 50-315/97201-15; 50-316/97201-15: Generic letter (GL) No. 89-13, "Service Water System Problems Affecting Safety-Related Equipment," testing of the CCW/ESW heat exchangers.

The licensee used a maximum fouling factor acceptance criteria of 0.00169 or less. This value was the maximum allowable fouling acceptable to remove the design heat load. This met the GL intent for an operating cycle, however, the AE team was concerned that this left no margin if fouling were to occur during the operating cycle. In addition, the AE team determined that the fouling factor acceptance criteria did not include instrument uncertainties. This is considered an unresolved item (URI 50-315/98009-15; URI 50-316/98009-15) pending further NRC review.

E8.16 (Closed) Unresolved Item 50-315/97201-16; 50-316/97201-16: Generic letter No. 89-13, "Service Water System Problems Affecting Safety-Related Equipment," testing of the emergency diesel generator heat exchangers.

Heat exchanger performance trending included the EDG jacket water, lube oil and aftercoolers. ESW outlet temperatures were recorded and trends were charted over several tests. The trends indicated that the temperature profiles were relatively constant over the testing period. However, the AE team identified that the heat exchanger outlet temperature was automatically regulated by a flow control valve. Therefore, the trending data only indicated that the flow control valves were operating correctly. This is considered an unresolved item (URI 50-315/98009-16; URI 50-316/98009-16) pending further NRC review.

E8.17 (Closed) Unresolved Item 50-315/97201-17; 50-316/97201-17: Inadequate justification to demonstrate Unit 2, 250 Vdc, CD battery operability.

Cell No. 34 was discovered reading less that the TS required 2.13 volts. Temporary modification No. 2-IHP-5021.EMP.009, dated June 19, 1997, installed an individual cell charger on cell No. 34. Subsequently, the cell voltage increased to 2.214 volts. The licensee performed an operability determination and concluded that the battery was operable. However, the cell voltage reading was taken with the individual charger installed. The AE team was concerned that the reading did not represent the true cell state-of-charge. The cell remained on an individual cell charger for 51 days. The cell was replaced on August 11, 1997. This is considered an unresolved item (URI 50-315/98009-17; URI 50-316/98009-17) pending further NRC review.

E8.18 (Closed) Unresolved Item 50-315/97201-18; 50-316/97201-18: The licensee did not always meet RWST, Appendix R, alternate borated water supply requirements.

The fire protection quality assurance program was described in letter No.

AEP:NRC:0847AE, dated August 1, 1997, “Quality Assurance Program Description (QAPD) Proposed Revision.” Section 1.7.19.1, Fire Protection QA Program - Introduction, stated, in part, that the Fire Protection QA Program encompasses design, procurement, fabrication, construction, surveillance, inspection, operation, maintenance, modification, and audits. In addition, Section 1.7.19.3, “Design Control and Procurement Document Control,” stated, in part, that design changes, including field changes and deviations, are controlled by procedures.

As of September 12,1997, calculation No. TH-90-02, dated February 20, 1990, “RCS Volume Make-up Required After Appendix R Fire,” determined that the borated water volume from the RWST should be increased from 30,629 to 87,000 gallons. However, procedure No. PMP-4100, dated February 20, 1996, ”Plant Shutdown Safety and Risk Management,” was not revised from 30,629 to 87,000 gallons. As a consequence, there were times when the RWST water volume was less than 87,000 gallons in the shutdown Unit RWST which did not meet the Appendix R alternate borated water supply requirement for the operating unit. The failure to correctly translate the RWST Appendix R inventory design basis into specifications, drawings, procedures, and instructions is considered an apparent violation (EEI 50-315/98009-18; EEI 50-316/98009-18) of 10 CFR 50, Appendix B, Criterion III, “Design Control.”

E8.19 (Closed) Unresolved Item 50-315/97201-19; 50-316/97201-19: CCW outlet temperature loop uncertainties.

The licensee could not provide a CCW heat exchanger outlet temperature loop uncertainty calculation. Only the high and low temperature alarm values were accounted for by ECP No. 1-2-C4-02. This is considered an unresolved item (URI 50-315/98009-19; URI 50-316/98009-19) pending further NRC review of the uncertainty measurement process.

E8.20 (Closed) Unresolved Item 50-315/97201-20; 50-316/97201-20: ESW intake temperature loop uncertainties.

The licensee performed a calculation and identified that the loop uncertainty error was ± 3.52°F. This is considered an unresolved item (URI 50-315/98009-20; URI 50-316/98009-20) pending further NRC review of the calculation.

E8.21 (Closed) Unresolved Item 50-315/97201-21; 50-316/97201-21: Control room temperature loop uncertainties.

The licensee performed a calculation and identified that the loop uncertainty error was ± 5.35°F. This is considered an unresolved item (URI 50-315/98009-21; URI 50-316/98009-21) pending further NRC review of the calculation.

E8.22 (Closed) Unresolved Item 50-315/97201-22; 50-316/97201-22: Setpoint control program weaknesses.

The AE team determined that the RHR and CCW instrumentation and control systems were adequately designed and installed. However, other setpoint control aspects need to be reviewed for potential impact on other systems. This is considered an unresolved item (URI 50-315/98009-22; URI 50-316/98009-22) pending further NRC review.

E8.23 (Closed) Unresolved Item 50-315/97201-23; 50-316/97201-23: Licensee considered changes to procedures in the conservative direction to be non-intent changes.

The non-intent procedure change process was permitted by TS 6.5.3.1a, “Technical Review and Control.” The AE team identified several procedures where process parameters were changed and implemented as non-intent procedure changes without performing a 10 CFR 50.59 screening. The licensee considered changes to procedure process parameters that were in the conservative direction to be non-intent changes. Therefore, the change could be implemented with only two signatures prior to receiving a formal review. The formal review was not required for 14 days. This is considered an unresolved item (URI 50-315/98009-23; URI 50-316/98009-23) pending further NRC review of the procedure change process.

E8.24 (Closed) Unresolved Item 50-315/97201-24; 50-316/97201-24: Plant drawings and design specifications did not conform to American Standard Code for Pressure Piping requirements (ANSI B31.1, 1967 edition).

The AE team identified the following Code deviations:

- CCW piping inside containment conflict with the piping specification and classification requirements.
- CCW system overpressure protection deviated from B31.1 requirements.
- RHR low pressure protection interlock protection was defeated during Mode 4 operation.
- CCW heat exchanger lacked overpressure protection.

This is considered an unresolved item (URI 50-315/98009-24; URI 50-316/98009-24) pending further NRC review.

E8.25 (Closed) Unresolved item 50-315/97201-25; 50-316/97201-15: Plant equipment abandoned in place without proper reviews and controls.

Licensee policy No. 227000-POL-5400-02, dated July 14, 1995, “Treatment of Abandoned in Place Items,” stated, in part, that items abandoned in place should typically be removed as part of the design change process. However, the AE team identified eight (8) pieces of plant equipment that were abandoned in place without following the policy statement. This is considered an unresolved item (URI 50-315/98009-25; URI 50-316/98009-25) pending further NRC review.

E8.26 (Closed) Unresolved Item 50-315/97201-26; 50-316/97201-26: Several design documentation discrepancies were identified.

The AE team identified the following discrepancies:

- Design calculations were not revised to account for higher ultimate heat sink (lake) temperatures.

- RHR calculation contained several non-conservative design inputs.

- RWST level instrument uncertainty calculation contained an elevation error and a pump flow error.

- Several drawing errors were identified.

This is considered an unresolved item (URI 50-315/98009-26; URI 50-316/98009-26) pending NRC review of these discrepancies.

E8.27 <u>(Closed) Unresolved Item 50-315/97201-27; 50-316/97201-27</u>: Potential for CCW flashing due to a procedure deficiency.

The AE team was concerned that calculation No. NEMP 960519AF, “CCW LOCA/Cooldown Analysis for U2 Uprating Program,” assumption had not been incorporated into operation procedure No. 01(02)-OHP 4021.001.004, “Plant Cooldown from Hot Standby to Cold Shutdown.” The potential existed for a water hammer and/or other damaging type transient to occur during cooldown. This is considered an unresolved item (URI 50-315/98009-27; URI 50-316/98009-27) pending further NRC review.

E8.28 <u>(Closed) Unresolved Item 50-315/97201-28; 50-316/97201-28</u>: Exceeding UFSAR stated ultimate heat sink (lake) temperatures.

The AE team identified as of September 10, 1997, that the units had been operated above the ESW 76°F temperature limit specified in UFSAR Tables 6.3-2 and 9.5-3. A 10 CFR 50.59 safety evaluation that fully analyzed plant operation above 76°F was not provided. Specifically, during July and August of 1988, the units were operated in an unanalyzed condition for 22 continuous days at an average ultimate heat sink temperature of 81°F. The failure to fully analyze plant operation above 76°F is considered an apparent violation (EEI 50-315/98009-28; EEI 50-316/98009-28) of 10 CFR 50.59, “Changes, Tests, and Experiments,” requirements.

E8.29 <u>(Closed) Unresolved Item 50-315/97201-29; 50-316/97201-29</u>: Unit 2 dual train CCW and ESW outage.

During the Unit 2 full core off-load outage in 1996 and with Unit 1 at 100% power, both Unit 2 CCW and ESW trains were taken out-of-service on August 7 through 8, 1996, leaving one Unit 1 CCW train available to supply spent fuel pool (SFP) cooling. Specifically, the March 11 and March 20, 1996, 50.59 safety evaluations performed for the core off-load did not recognize that the Unit 1 CCW system could not perform its safety function under the design basis assumptions described in the UFSAR. Specifically, a single CCW train operating at 95°F could not maintain the SFP bulk water temperature less than specified (160°F) in UFSAR Section 9.4, “Spent Fuel Pool Cooling System.” In addition, with a single Unit 1 CCW train providing SFP cooling, a Unit 1 design basis accident would isolate CCW causing the loss of SFP cooling. As a consequence, the SFP time-to-boil margin would be reduced. The safety evaluations failed to consider SFP cooling loss during a design basis accident on Unit 1 and the resulting reduction in time-to-boil margin during the Unit 2 dual train CCW and ESW outage. This is considered an apparent violation (EEI 50-315/98009-29; EEI 50-316/98009-29) of 10 CFR 50.59, “Changes, Tests, and Experiments, “ requirements.

E8.30 (Closed) Unresolved Item 50-315/97201-30; 50-316/97201-30: Emergency operating procedure Nos. 01(02)-OHP 4023.ES-1.3, Revision 2, “Transfer to Cold Leg Recirculation,” procedure change created a single failure vulnerability.

Procedure ES-1.3 was revised (Revision 2) in June 1992 to piggy-back both centrifugal charging and safety injection trains onto the west RHR pump. However, the 10 CFR 50.59 safety evaluation for this procedure revision did not identity that UFSAR Section 6.2, “Emergency Core Cooling Systems,” in use in 1992, required that the operator first transfer one ECCS train to recirculation and then transfer the other ECCS train. As a consequence, a failure of the west RHR pump would cause the loss of all high head emergency core cooling. In addition, procedure ES-1.3, Revision Nos. 3 and 4, and their corresponding safety evaluations did not identify the single failure vulnerability and that ES-1.3 incorrectly implemented the UFSAR described transfer sequence from injection to recirculation. The safety evaluations were inadequate by failing to identify the single failure vulnerability. This is considered an apparent violation (EEI 50-315/98009-30; EEI 50-316/98009-30) of 10 CFR 50.59, “Changes, Tests, and Experiments,” requirements.

E8.31 (Closed) Unresolved Item 50-315/97201-31; 50-316/97201-31: Recirculation sump roof vent hole design basis not understood.

The NRC resident staff questioned the licensee as to the purpose for the ¾ inch recirculation sump roof vent holes. The design basis for the vent holes could not be determined since they did not appear on flow diagrams and were not discussed in the UFSAR. The vent holes were subsequently plugged in 1996 for Unit 2 and 1997 for Unit 1. The design control process was not utilized since the licensee believed the repair was returning the sump roof to its original design. However, the holes were described in letter No. AEP:NRC:00110, dated December 29, 1978, committing to install the vent holes. As a consequence, the design control process was bypassed and the UFSAR was not updated with the containment sump roof vent hole design basis. The failure to correctly translate the ¾ inch containment recirculation sump roof vent hole design basis into specifications, drawings, procedures, and instructions is considered an apparent violation (EEI 50-315/98009-31; EEI 50-316/98009-31) of 10 CFR 50, Appendix B, Criterion III, “Design Control.”

E8.32 (Closed) Unresolved Item 50-315/97201-32; 50-316/97201-32: Exceeding UFSAR stated CCW operating temperatures.

The AE team identified as of September 10, 1997, that both units had been operated with CCW temperatures (120°F) above UFSAR Table 9.5-3 specified design value of 95°F. As a consequence, the potential existed for the reactor coolant pump (RCP) seals to fail. A 50.59 safety evaluation had not been performed to review this unanalyzed condition. The failure to perform a safety evaluation for exceeding a UFSAR stated design value is considered an apparent violation (EEI 50-315/98009-32; EEI 50-316/98009-32) of 10 CFR 50.59, “Changes, Tests, and Experiments,” requirements.

E8.33 (Closed) Unresolved Item 50-315/97201-33; 50-316/97201-33: Unit operation with RCP thermal barrier CCW flow less than the UFSAR value.

The AE team identified as of September 10, 1997, that both units had been operated with RCP thermal barrier CCW flows between 25 and 35 gpm. However, UFSAR Table 9.5-2 stated, in part, that the minimum flow was 35 gpm to each thermal barrier. A 50.59 safety evaluation had not been performed to review this unanalyzed condition, specifically, for flows less than 28 gpm. The failure to perform a safety evaluation for exceeding a UFSAR stated design value is considered an apparent violation (EEI 50-315/98009-33; EEI 50-316/98009-33) of 10 CFR 50.59, "Changes, Tests, and Experiments," requirements.

E8.34 (Closed) Unresolved Item 50-315/97201-34; 50-316/97201-34: Operation of the RHR system without overpressure protection.

The licensee identified on September 11, 1997, that both units had been operated with RHR overpressure protection that did not meet UFSAR requirements. Specifically, this operating practice did not meet the assumptions identified in UFSAR Section 9.3, "Residual Heat Removal System." The RHR overpressure interlock associated with RHR hot leg inlet isolation valve Nos. ICM-129 and IMO-128 was defeated without performing a 50.59 safety evaluation for an operating practice that differed from the UFSAR. The purpose of the interlock was to prevent the operators from opening the valves when the reactor coolant system (RCS) pressure was above 400 psig and to provide automatic valve closure when RCS pressure exceeded 600 psig. However, when operating in Mode 4, power was removed from these valves to prevent spurious closure during shutdown cooling operation. This defeated the automatic closure feature as described in the UFSAR. The failure to perform a safety evaluation for an operating practice that differed from the UFSAR is considered an apparent violation (EEI 50-315/98009-34; EEI 50-316/98009-34) of 10 CFR 50.59, "Changes, Tests, and Experiments," requirements.

E8.35 (Closed) Information Followup Item 50-315/97201-01; 50-316/97201-01: UFSAR and TS RWST volume inconsistencies.

The AE team did not clearly understand the RWST water volume design basis. UFSAR Section 6.2.2 stated, in part, that the RWST was maintained with a minimum volume of 350,000 gallons of borated water above the bottom of the RWST discharge pipe. TS 3/4.5.5 stated, in part, that the maintained minimum volume of RWST borated water was 350,000 gallons. This is considered an information followup item (IFI 50-315/98009-35; IFI 50-316/98009-35) pending further NRC review.

E8.36 (Closed) Information Followup Item 50-315/97201-02; 50-316/97201-02): ECCS level instrumentation and equipment allowed out-of-service times.

The licensee relies on RWST and containment sump level instrumentation during ECCS pump suction transfer from the RWST to the containment recirculation sump. The out-of-service time for ECCS equipment was defined in TS as 72 hours. However, the out-of-service time for the level instrumentation was defined in TS as 30 days. The AE team was concerned that the instrumentation out-of-service time was not commensurate with the ECCS out-of-service time. This is considered an information followup item (IFI 50-315/98009-36; IFI 50-316/98009-02) pending further NRC review.

V. Management Meetings

X1 Exit Meeting Summary

On April 15, 1998, the inspectors presented the inspection results to licensee management. The licensee acknowledged the findings presented.

The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

K. Baker, Production Engineering
A. Blind, Nuclear Engineering Vice President
D. Hafer, Plant Engineering
J. Kingseed, Nuclear Safety and Analysis
T. Postlewait, Design Engineering
J. Sampson, Site Vice President

NRC

J. Gavula, Chief, Engineering Specialists Branch 1, RIII
J. Maynen, Cook Resident Inspector

INSPECTION PROCEDURE USED

IP 92903	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

All identified items apply to Docket Nos. 50-315; 50-316

Closed (NRR)	Opened (RIII)	Description
URI 97201-01	EEI 98009-01	RWST Level Instrumentation Bias Errors
URI 97201-02	URI 98009-02	RWST Level Instrument Uncertainties
URI 97201-03	EEI 98009-03	RWST Low-Low Level Setpoint
URI 97201-04	URI 98009-04	Vortexing Issue Corrective Actions
URI 97201-05	EEI 98009-05	Containment Sump Level Uncertainties
URI 97201-06	EEI 98009-06	Containment Sump Design Basis
URI 97201-07	URI 98009-07	Single Active Failure Definition
URI 97201-08	EEI 98009-08	Sump ¼ inch Particulate Retention Design Basis
URI 97201-09	URI 98009-09	Valve Leak Testing
URI 97201-10	URI 98009-10	One Train CCW Cooldown
URI 97201-11	URI 98009-11	CCW Heat Exchanger Modeling Error
URI 97201-12	EEI 98009-12	Control Room Temperature Evaluation
URI 97201-13	URI 98009-13	Instrument Air System Failure Modes
URI 97201-14	EEI 98009-14	CCW Heat Exchanger Flow Exceeds UFSAR Value
URI 97201-15	URI 98009-15	CCW/ESW Heat Exchanger Testing
URI 97201-16	URI 98009-16	EDG Heat Exchanger Testing
URI 97201-17	URI 98009-17	250 Vdc Battery Single Cell Charging
URI 97201-18	EEI 98009-18	Appendix R Alternate Borated Water Supply
URI 97201-19	URI 98009-19	CCW Outlet Temperature Loop Uncertainties
URI 97201-20	URI 98009-20	ESW Intake Temperature Loop Uncertainties
URI 97201-21	URI 98009-21	Control Room Temperature Loop Uncertainties
URI 97201-22	URI 98009-22	Setpoint Control Program
URI 97201-23	URI 98009-23	Non-intent Procedure Changes
URI 97201-24	URI 98009-24	Piping Code Deviations
URI 97201-25	URI 98009-25	Abandoned Plant Equipment
URI 97201-26	URI 98009-26	Design Document Discrepancies
URI 97201-27	URI 98009-27	Potential For CCW Flashing
URI 97201-28	EEI 98009-28	Exceeding Ultimate Heat Sink Temperatures
URI 97201-29	EEI 98009-29	Unit 2 Dual Train CCW/ESW Outage
URI 97201-30	EEI 98009-30	EOP ES-1.3 Single Failure Vulnerability
URI 97201-31	EEI 98009-31	Sump Roof Vent Hole Design Basis
URI 97201-32	EEI 98009-32	Exceeding CCW Design Temperature
URI 97201-33	EEI 98009-33	RCP Thermal Barrier CCW Flow Low
URI 97201-34	EEI 98009-34	RHR Overpressure Protection
IFI 97201-01	IFI 98009-35	UFSAR And TS RWST Volume Inconsistencies
IFI 97201-02	IFI 98009-36	RWST Level Instrument Out-Of-Service Times

LIST OF ACRONYMS USED

AE	Architectural and Engineering
AEP	American Electric Power
ATWS	Anticipated Transient Without Scram
CAL	Confirmatory Action Letter
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CREVS	Control Room Emergency Ventilation System
CR	Condition Report
CTS	Containment Spray System
DBA	Design Basis Accident
DCN	Design Change Notice
DCP	Design Change Package
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
ECP	Engineering Control Procedure
EDG	Emergency Diesel Generator
EEI	Escalated Enforcement Item
EOP	Emergency Operating Procedure
ESW	Essential Service Water
°F	Degree Fahrenheit
gpm	gallons per minute
IFI	Information Follow up Item
LER	Licensee Event Report
LOCA	Loss-Of-Coolant-Accident
MWt	Mega-Watt thermal
NPSH	Net Positive Suction Head
PMI	Plant Manager Instruction
PMP	Plant Manager Procedure
psig	Pounds Per Square Inch Gauge
QA	Quality Assurance
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFC	Request For Change
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SFP	Spent Fuel Pool
SI	Safety Injection
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
W	Westinghouse